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REACTOR PRESSURE VESSEL TEMPERATURE ANALYSIS OF CANDIDATE VERY HIGH TEMPERATURE REACTOR DESIGNS

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ABSTRACT

Analyses were performed to determine maximum temperatures in the reactor pressure vessel for two potential Very-High Temperature Reactor (VHTR) designs during normal operation and during a depressurized conduction cooldown accident. The purpose of the analyses was to aid in the determination of appropriate reactor vessel materials for the VHTR. The designs evaluated utilized both prismatic and pebble-bed cores that generated 600 MW of thermal power. Calculations were performed for fluid outlet temperatures of 900 and 950 °C, corresponding to the expected range for the VHTR. The analyses were performed using the RELAP5-3D and PEBBED-THERMIX computer codes. Results of the calculations were compared with preliminary temperature limits derived from the ASME pressure vessel code.

Because PEBBED-THERMIX has not been extensively validated, confirmatory calculations were also performed with RELAP5-3D for the pebble-bed design. During normal operation, the predicted axial profiles in reactor vessel temperature were similar with both codes and the predicted maximum values were within 2 °C. The trends of the calculated vessel temperatures were similar during the depressurized conduction cooldown accident. The maximum value predicted with RELAP5-3D during the depressurized conduction cooldown accident was about 40 °C higher than that predicted with PEBBED. This agreement is considered reasonable based on the expected uncertainty in either calculation. The differences between the PEBBED and RELAP5-3D calculations were not large enough to affect conclusions concerning comparisons between calculated and allowed maximum temperatures during normal operation and the depressurized conduction cooldown accident.

INTRODUCTION

The Idaho National Laboratory is engaged in research and development toward the construction of a Very High Temperature Reactor (VHTR) employing TRISO fuel in either a prismatic or pebble-bed core configuration. A critical design limitation is the peak temperature of the reactor pressure vessel (RPV) during normal operation and during severe loss of flow accidents. The expected high operating temperatures of the Very High Temperature Reactor (VHTR) place significant constraints on the choice of materials for the reactor pressure vessel. Current designs based on the Pebble Bed Modular Reactor (PBMR) propose using SA-508 steel for the reactor vessel while those based on the Gas-Turbine Modular Helium Reactor (GT-MHR) propose using SA-336 (9Cr-1Mo-V). SA-508 steel [1] has been used for reactor vessels in light water reactors and is expected to be significantly cheaper than SA-336.

For this analysis, prototypical 600 MWt prismatic and pebble-bed core designs operating with a 900 °C outlet temperature were modeled to obtain estimates of RPV temperature profiles under steady state and severe accident conditions. The prismatic reactor design is based upon the General Atomics GT-MHR but with coolant inlet and outlet temperatures raised per the requirements of the VHTR. The inlet coolant path was also modified to accommodate the higher temperatures and the material limitations of the 9Cr-1Mo-V steel alloy that is proposed for use in this application. The pebble-bed reactor was designed using the PEBBED code which features an automated genetic algorithm search routine to optimize the reactor geometry to meet user-specified required. Core and reflector dimensions were chosen to minimize the core height and pressure vessel diameter while keeping peak fuel temperatures below 1600 °C during a complete depressurized loss of coolant accident.

Analyses were performed to determine maximum temperatures in the reactor pressure vessel for two potential VHTR designs during normal operation and during a depressurized conduction cooldown accident. The purpose of the analyses was to aid in the determination of appropriate reactor vessel materials for the VHTR. The VHTR designs evaluated utilized both prismatic and pebble-bed cores. The cores generated 600 MW of thermal power and had fluid outlet temperatures of either 900 or 950 °C. The analyses of the prismatic design were performed using the RELAP5-3D computer code, while the analyses of the pebble-bed design were performed with the coupled PEBBED-THERMIX.

NOMENCLATURE

ASME	American Society of Mechanical Engineers
BAF	Bottom of Active Fuel (Core)
B_4C	boron carbide
GT-MHR	Gas-Turbine Modular Helium Reactor
He	helium
MW	megawatt
MPa	megapascal
OECD	Organization for Economic Cooperation and Development
PBMR	Pebble Bed Modular Reactor
RCCS	Reactor Cavity Cooling System
RPV	Reactor Pressure Vessel
TAF	Top of Active Fuel (Core)
VHTR	Very High Temperature Reactor

1.0 Limits on Pressure Vessel Materials

Table I summarizes the results of the calculations with preliminary temperature limits from the ASME pressure vessel code [2]. The temperature limits are considered preliminary because not all factors, such as reductions to account for welds and operating history, have been accounted for. Calculated results for normal operation are presented in the second or third columns of Table 1. Calculated results for the depressurized conduction cooldown accident are presented in the last column.

Table 1. Temperature limits for SA-508 and SA-336 steels.

Steel	Operation			
	Unlimited	< 3x10^5 h	< 3000 h	< 1000 h
SA-508	T < 371 °C		371 < T < 427 °C	427 < T < 538 °C
SA-336 (9Cr-1Mo-V)	T < 371 °C	371 < T < 590 °C		590 < T < 650°C

Table I shows that SA-508 is not a suitable material for the prismatic VHTR based on its current design. The maximum temperature during normal operation exceeds the value that allows unlimited operating time with SA-508 steel. The calculations show that SA-336 steel is suitable for the prismatic design and allows 34 years of normal operation with considerable margin. Additional work will be required to extend the design lifetime to the 40 to 60 years typically desired for nuclear plants. A more detailed design of the outlet plenum region is also required. The current design provides a large margin to the 1000-h limit of 650 °C for SA-336 during the depressurized conduction cooldown accident. In fact, the maximum calculated temperatures during the accident meet the 590-°C limit for normal operation.

For the pebble-bed design, the maximum vessel temperature during normal operation was 14 to 30 °C below the allowed value for unlimited operation of SA-508. Some design changes, such as the use of active cooling of the lower vessel, may be required to increase the margin to the temperature limit during normal operation. The maximum calculated temperatures were significantly below the allowed values during the depressurized conduction cooldown accident.

A pressurized loss of flow condition can also raise pressure vessel temperatures to excessively high values. A study by MacDonald [3] et al. indicates that the temperatures achieved in the reactor pressure vessel can exceed those attained in the depressurized case because of the stronger coupling between the core and external structures. The differences, however, are small enough (<50 °C) such that the conclusions of this study are not invalidated. For simplicity, only the depressurized accident was investigated in this study.

The RELAP5-3D and PEBBED-THERMIX computer codes are described in Section 2. Results of the thermal-hydraulic analyses of the prismatic and pebble-bed designs are presented in Section 3. Conclusions are presented in Section 4. References are presented in Section 5.

2.0 Code Description

2.1 RELAP5-3D

RELAP5-3D (RELAP5-3D Code Development Team 2005) was developed for the thermal-hydraulic analysis of light water reactors and related experimental systems during loss-of-coolant accidents and operational transients. The code originally contained a one-dimensional, nonhomogeneous, and nonequilbirum model for two-phase flow and a point model for reactor kinetics. The code currently contains multi-dimensional models for flow and reactor kinetics, but one-dimensional models were used in this analysis.

RELAP5-3D has the capability to simulate a wide variety of nuclear systems and working fluids. The code currently simulates over twenty working fluids, including light and heavy water, helium, carbon dioxide, hydrogen, nitrogen, helium-xenon, lead-bismuth, sodium, lithium, potassium, sodium-potassium, ammonia, and several liquid salts.

Recent assessments of RELAP5-3D that are applicable to the VHTR are described by Oh et al. [4].

A special version of the RELAP5-3D was created for the pebble-bed analysis described in Section 3.2. The special code version implemented correlations for heat transfer and wall friction coefficients applicable for pebble-bed reactors based on correlations given by Fenech [5].

2.2 PEBBED-THERMIX

PEBBED is a three-dimensional core simulator code developed at the Idaho National Laboratory specifically for pebble-bed reactor design and analysis [6,7]. It converges directly upon the asymptotic or equilibrium fuel cycle using an integrated neutron diffusion-depletion solver. Core design optimization is performed using a genetic algorithm operating on core geometry and pebble flow parameters. Core temperature profiles can be obtained using a one-dimensional embedded thermal-hydraulic solver or with a coupled THERMIX calculation. THERMIX is a two-dimensional (RZ) heat transfer and gas dynamics code developed for the German pebble-bed reactor program. It is part of the VSOP PBR fuel cycle analysis system [8] but the THERMIX module has been extracted recently for use with other codes. THERMIX contains material properties and correlations validated for PBR analysis under the German High Temperature Reactor program. The PBMR Corporation uses a modified version of VSOP (with THERMIX) for their reactor design and fuel cycle analyses [9].

Burnup and temperature feedback are treated by iterating between PEBBED-THERMIX and COMBINE, the INL spectrum code [10].

3.0 Results

Results of the thermal-hydraulic analyses of the prismatic and pebble-bed versions of the VHTR are presented in Sections 3.1 and 3.2, respectively.

3.1 Prismatic Design

The evaluation of the VHTR containing prismatic blocks is based on the design of the GT-MHR [11]. The GT-MHR was designed to operate at core inlet and outlet temperatures of 491/850 °C. The inside wall temperature of the reactor vessel was 485 °C, which was just slightly below the core inlet fluid temperature. The reactor vessel was fabricated from 9Cr-1Mo-V ferritic steel. The vessel design limits used by General Atomics were 495 °C for 4.6 x 10^5 h (normal operation) and 538 °C for 1.0 x 10^3 h (transient conditions). General Atomics reported a maximum reactor vessel (midwall) temperature of 490 °C versus an accident limit of 565 °C during a depressurized conduction cooldown accident. Note that the temperature limits used by General Atomics predated the ASME code for 9Cr-1Mo-V steel and were about 100 °C lower than the values given in Table 1.

The VHTR is desired to operate at a higher outlet temperature than the GT-MHR to increase the efficiency of the electrical production cycle and/or to allow for efficient production of hydrogen. Although the VHTR has not yet been designed, current goals call for outlet temperatures in the range of 900 to 950 °C. Raising the GT-MHR inlet and outlet temperatures by 50 to 100 °C would achieve the desired temperature goal for efficiency, but would also increase the maximum reactor vessel temperature. Maintaining the inlet temperature at the GT-MHR value and reducing the core flow to achieve the desired outlet temperature would keep the vessel temperature relatively low but would cause concerns about flow starvation in the hot channels of the core.

Reza, et al. [12] evaluated design modifications to the GT-MHR that routed the inlet flow through holes in the outer reflector rather than through channels between the core barrel and the reactor vessel. This design prevented the inlet flow from contacting the reactor vessel, which enabled the inlet and outlet fluid temperatures to be increased while lowering the operating temperature of the reactor vessel. Detailed simulations of the vessel lower head were not performed, but it was assumed that the lower head could be isolated from the inlet flow, such as by applying insulation or by using a distribution header to supply the flow holes in the outer reflector.

This evaluation utilized the GT-MHR design as modified by Reza et al. [12]. RELAP5-3D [12] calculations were performed to determine the reactor vessel temperature during normal operation and a depressurized conduction cooldown, or depressurized loss of flow, accident. The RELAP5-3D input model was based on the model of Reza which in turn was based on the model developed for the Next Generation Nuclear Plant by MacDonald et al. [3]. The model of Reza et al. was modified to produce the desired core inlet/outlet temperature conditions for this evaluation.

The RELAP5-3D model of the VHTR vessel is shown in Figure 1. The active core (Components 152 through 156) was modeled with three radial rings, corresponding to the three annular rings of prismatic fuel blocks in the design, and ten axial levels. The upper and lower reflectors were each modeled with a single axial level. Core bypass paths were simulated through the inner and outer reflectors (Components 142 and 145, respectively). In the original model developed by MacDonald et al. [3], cold helium from the vessel inlet flowed upwards through an annular riser (Component 130) and into the core inlet plenum (Component 140). In the revised model developed by Reza et al., the riser was modeled as a region of stagnant helium and the cold helium flowed upwards through flow holes in the outer reflector (Component 132). Conduction enclosure models were used to simulate the heat transfer between the three core rings and between the core and the reflectors. Radiation enclosure models were used to simulate the heat transfer across the riser and between the reactor vessel and the containment wall. The outer surface of the containment wall was cooled by the reactor cavity cooling system (RCCS). Naturally-circulating air was the ultimate heat sink for the RCCS.

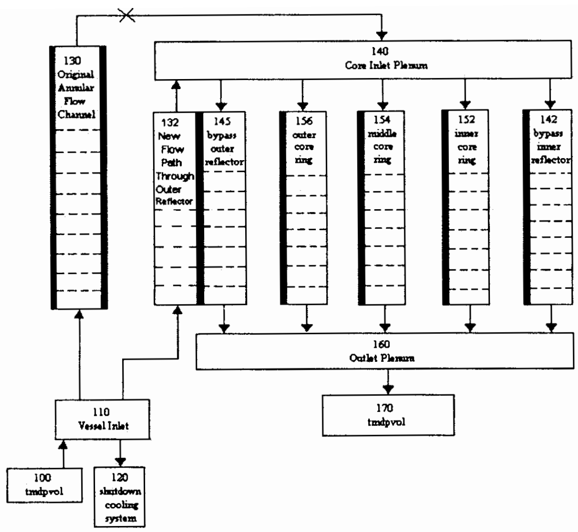

Figure 1: RELAP5-3D model of the VHTR reactor vessel (from Reza et al. [12].

Steady-state calculations were performed with vessel outlet fluid temperatures of 900 and 950 °C. Results are presented in Table 2. The table contains two vessel temperatures, one at the inner wall and the other at the radial center of the inner and outer walls. Both maximum temperatures generally occurred at the same axial location in the vessel. The midwall temperature is appropriate for comparison with the temperature limits given in Table 1 because the ASME code (2004) specifies the use of the wall-averaged temperature. The higher temperature at the inner wall is an indication of the heat conduction through the wall and the subsequent heat transfer to the RCCS.

Table 2. Calculated thermal-hydraulic conditions during normal operation for the prismatic VHTR.

Parameter	T_{out}=900 °C	T_{out}=950 °C
Power, MW	600	600
Pressure, MPa	7.00	7.00
Differential pressure, MPa	0.0764	0.0802
Inlet temperature, °C	540	590
Outlet temperature, °C	900	950
Flow rate, kg/s	325.0	325.1
Core bypass, %	9.59	9.53
Maximum vessel temperature (inner wall), °C	410	447
Maximum vessel temperature (midwall), °C	388	421
Maximum fuel temperature, °C	1064	1112
RCCS power, MW	1.83	2.13

The steady-state reactor vessel temperature was far less than the limit of 590 °C for normal operation with SA-336 steel given in Table 2. The calculated results are expected to be reasonably accurate because the original RELAP5-3D model developed by MacDonald et al. [3] was benchmarked against results calculated by General Atomics for the GT-MHR. The benchmarking showed that the maximum reactor vessel temperatures predicted by RELAP5-3D during normal operation and during the depressurized conduction cooldown accident were within 5 and 13 °C, respectively, of the values reported by General Atomics.

The calculated response of the maximum midwall temperature of the reactor vessel during a depressurized conduction cooldown accident is shown in Figure 2. During normal operation, heat transfer due to convection to the fluid in the gaps between reflector blocks and the coolant channels in the outer reflector was much greater than the heat transfer due to conduction through the reflector. Conduction became the dominant heat transfer mechanism after the scram and the depressurization. The temperatures of the core barrel and the reactor vessel initially decreased during the accident due to the decrease in convective heat transfer combined with the long thermal response time of the outer reflector. Eventually the temperature gradients due to conduction worked their way through the outer reflector and the vessel temperature increased because of the mismatch between the core decay power and the heat removed by the RCCS (see Figure 3). The reactor vessel temperature peaked near the time when the power removed by the RCCS exceeded the decay heat.

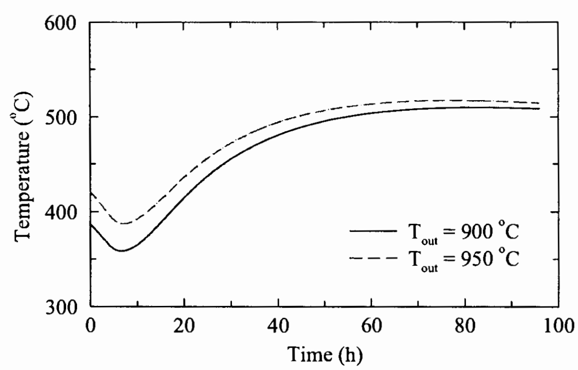

Figure 2. Maximum reactor vessel temperatures during a depressurized conduction cooldown accident with the prismatic design.

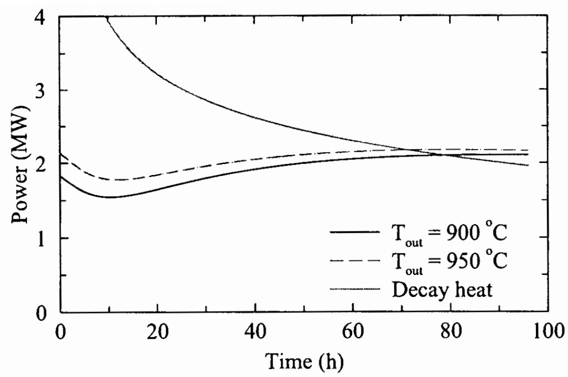

Figure 3. A comparison of RCCS heat removal and core decay power during a depressurized conduction cooldown accident with the prismatic design.

Table 3 presents maximum calculated temperatures during the conduction cooldown event. The maximum predicted midwall temperatures remained well below the accident limit of 650 °C given in Table 1. A comparison of Tables 2 and 3 reveals that a 50 °C increase in outlet temperature caused the maximum midwall temperature to increase by 33 °C at steady state, but by only 7 °C during the accident.

Table 3. Maximum temperatures during the depressurized conduction cooldown accident in the prismatic VHTR.

Parameter	T_{out} = 900 °C	T_{out} = 950 °C
Maximum vessel temperature (inner wall), °C	553	562
Maximum vessel temperature (midwall), °C	510	517
Maximum fuel temperature, °C	1501	1526

Figure 4 shows the axial temperature profiles at the inner surface of the reactor vessel at two times, 0.0 h which corresponds to normal, full-power operation and 83.3 h which is near the time of the peak temperature during the depressurized conduction cooldown accident. The reactor vessel temperature during normal operation is nearly independent of elevation in the region between the bottom of the active fuel (BAF) and the top of the active fuel (TAF). The wall temperature decreased above the active fuel because of the presence of a thermal shield. The calculated temperature below the active core decreased because the model assumed that the hot outlet fluid did not contact the vessel and that the only source of heat transfer to the lower head was due to radiation from adjacent heat structures. A more detailed evaluation of this potentially limiting region will be required once the design is completed.

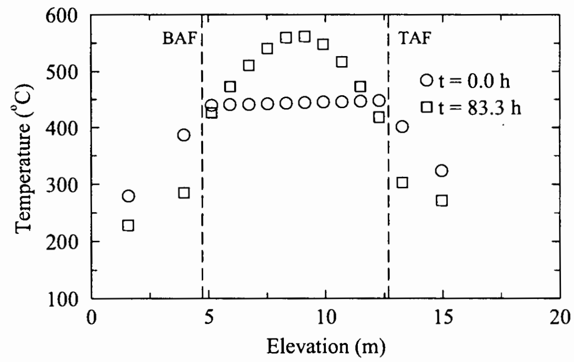

Figure 4. Reactor vessel axial temperature profile for the prismatic design with an outlet fluid temperature of 950 °C.

The axial temperature profile shifted significantly during the depressurized conduction cooldown accident. At the time of the peak temperature, the maximum vessel temperature occurred near the axial center of the core, which was the peak power location in this analysis.

3.2 Pebble-bed Design

The evaluation of the pebble-bed VHTR was based on PEBBED-THERMIX and RELAP5-3D models developed specifically for this task.

The PBMR has a rated thermal power of 400 MW and thus its basic design had to be modified to accommodate the higher power of the VHTR. A genetic algorithm was employed to automatically search for a core geometry that would yield a vessel comparable in size to other 600 MW high temperature reactor designs and still achieve passive safety, i.e. the fuel temperature will remain under 1600 °C during a depressurized conduction cooldown accident. Using the PBMR-400 design as a starting point, various perturbations to the core and reflector dimensions were explored and analyzed using PEBBED-THERMIX. Once a promising candidate was identified, final iterations with COMBINE were executed to generate a self-consistent physical model. PEBBED was also used to design a 600 MW pebble-bed VHTR in a previous study [3] but without detailed THERMIX analysis and temperature feedback in the cross sections. The reactor obtained in this current work is thus a more realistic model. The basic reactor dimensions are shown in Table 4.

Table 4. Dimensions of 600 MW pebble-bed VHTR obtained using a genetic algorithm search.

Dimension	Value (m)
Height of active core	10.96
Radius of inner reflector	1.582
Outer radius of fuel annulus	2.464
Outer radius of outer reflector	3.072

The thicknesses of the top and bottom axial reflectors (1.86 m and 4.36 m) were not changed from the values used in the PBMR-400 Coupled-Code Benchmark model [14]. Control rods (B_4C) were assumed to be partially inserted into the outer reflector to allow for xenon override. A complete description of the THERMIX model is shown in Figure 5.

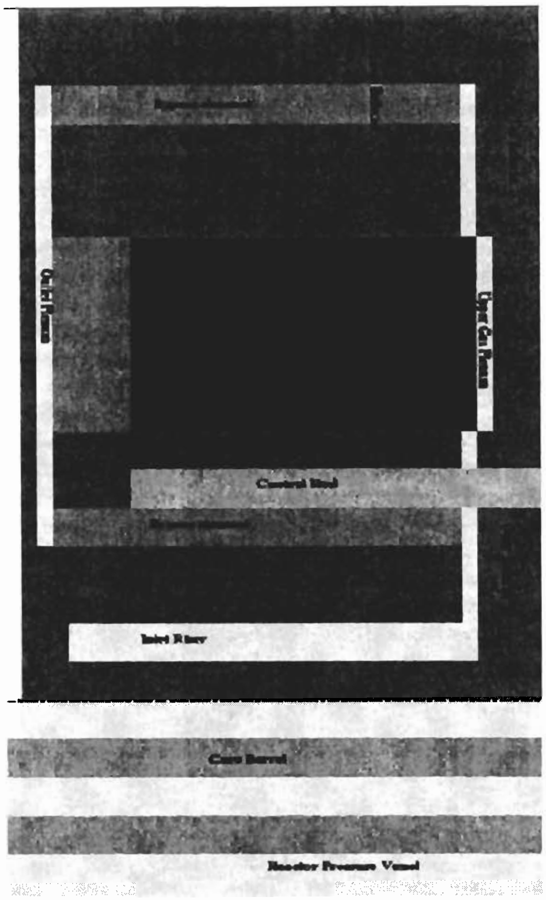

Figure 5: THERMIX model of 600 MW Pebble-bed VHTR

The active core is represented by composition 1. The cold He inlet is composition 14 and the coolant ascends through a riser (composition 15). It then passes through the upper plenum (composition 16), through the core and porous bottom reflector (composition 17) and out through the lower plenum (composition 18). Parts of the inner and outer radial reflectors are treated as porous to allow for bypass flow (composition 22). The pressure drops across these channels were adjusted to yield an overall bypass flow of 10%.

The upper and lower boundaries (compositions 20 and 21) are fixed as adiabatic so that heat transfer out of the vessel is through the radial boundary. This boundary is set at a constant temperature of 20 °C to simulate a working RCCS. The reactor pressure vessel is represented by composition 12.

The corresponding RELAP5-3D input model is shown in Figure 6. The heat source term was obtained from the PEBBED calculation and assumed that all fission power was deposited locally. The primary components are the riser (Component 120), gas plenum (Component 140), core (Components 151 through 155), and the outlet plenum (Component 180). The active core was modeled with five radial rings and eleven axial levels. Core bypass paths were simulated through the inner and outer reflectors (Components 142 and 145, respectively). The flow areas of these paths were adjusted to achieve a total bypass flow near 10%. The stagnant helium gaps between the outer reflector and the core barrel and the core barrel and the reactor vessel were modeled with Components 182 and 184. Conduction enclosure models were used to simulate the heat transfer between the various core rings and between the core and the reflectors. Radiation enclosure models were used to simulate the heat transfer across the stagnant gaps and between the reactor vessel and the containment wall. As with the THERMIX model, the outer surface of the containment wall was held at 20 °C, simulating a water-cooled RCCS. The outer surfaces of the top and bottom barrel plates, which are located above the gas plenum and below the outlet plenum, respectively, were modeled adiabatically.

Steady-state calculations were performed with outlet temperatures of 900 and 950 °C. Results are presented in Table 5.

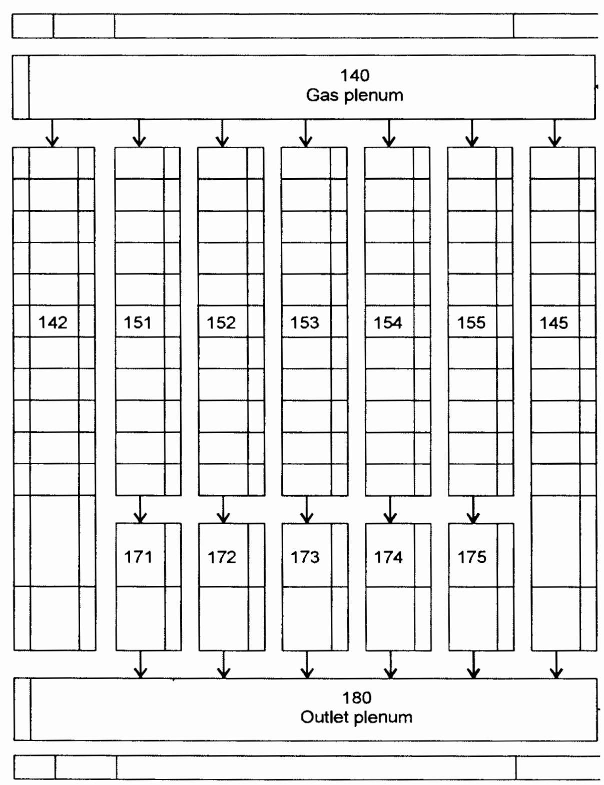

Figure 6. RELAP5-3D model of the pebble-bed design.

Table 5. RELAP5-3D and THERMIX calculated thermal-hydraulic conditions during normal operation for the pebble-bed VHTR.

Parameter	$T_{out} = 900$ °C		$T_{out} = 950$ °C	
	R5-3D	THRMX	R5-3D	THRMX
Power, MW	600	600	600	600
Pressure, MPa	9.00	9.00	9.00	9.00
Differential pressure, MPa	0.290	0.216	0.238	0.181
Inlet temperature, °C	482	tkm	482	482
Outlet temperature, °C	900	899	950	948
Flow rate, kg/s	275	276	246	247
Core bypass, %	9.8	10.1	10.0	9.9
Maximum vessel temperature (inner wall), °C	354	396	368	410
Maximum vessel temperature (midwall), °C	342	374	356	393
Maximum fuel temperature, °C	1072	1043	1141	1096
RCCS power, MW	1.43		1.46	

The steady states calculated with both codes were generally in excellent agreement. In particular, the calculations of vessel wall temperatures agreed within 8 °C. The differential pressure drop across the vessel was about 30% larger with RELAP5-3D, with the difference primarily occurring in the riser component. The maximum fuel temperatures were also somewhat higher with RELAP5-3D, probably because it explicitly accounted for the temperature rise between the surface and the center of a representative pebble in each core ring.

Several differences were noted between the steady states obtained for the pebble bed and those obtained previously with the prismatic core. First, the assumed inlet fluid temperature was lower with the pebble bed, which resulted in a larger temperature rise across the vessel and a lower mass flow rate. Second, even though the mass flow was reduced, the pressure drop across the vessel was larger because of the increased friction associated with the porous media compared to the open flow channels in the prismatic design. Third, the maximum vessel temperature was about 80 °C lower with the pebble bed, primarily because of the reduced sink temperature associated with the water-cooled RCCS versus the air-cooled design used with the prismatic core. Finally, the maximum fuel temperature in the pebble bed was about 30 °C higher than in the prismatic design. This result was unexpected because of the increased heat transfer coefficient from the pebbles compared to that from the coolant channels in the prismatic design. The cause of the higher fuel temperature was an increased maximum radial power peaking factor, 1.20 from PEBBED-THERMIX calculations versus 1.12 from the power profile assumed by MacDonald et al. [3] for the prismatic core.

The response of the pebble-bed VHTR as calculated by RELAP5-3D and THERMIX during a depressurized conduction cooldown accident is shown in the next several figures. Figure 7 shows the THERMIX-computed axial temperature profile at the inner surface of the reactor vessel at various times (in hours) during the accident. As was observed with the prismatic core, the maximum temperature initially decreased following the reactor scram and the decrease in convective heat transfer following the depressurization (see Figure 8). The maximum vessel temperature eventually increased because of the mismatch between the core decay power and the heat removed by the RCCS (see Figure 9). The sharp change in slope near ten hours occurred as the location of the maximum temperature switched from near the bottom of the vessel to near the axial center of the core. The maximum temperature remained far below the accident limit of 538 °C for SA-508 steel.

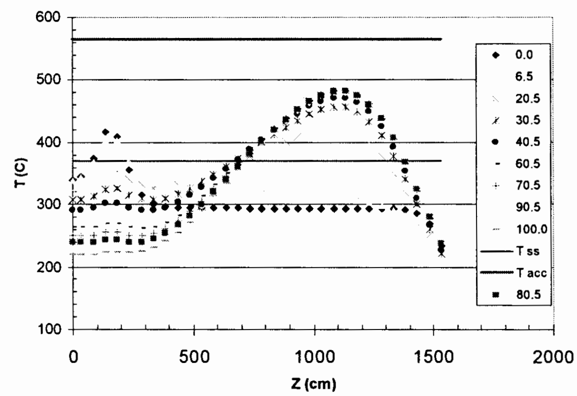

Figure 7. Reactor pressure vessel inner wall temperature profile during a depressurized conduction cooldown accident (950 °C outlet fluid temperature.)

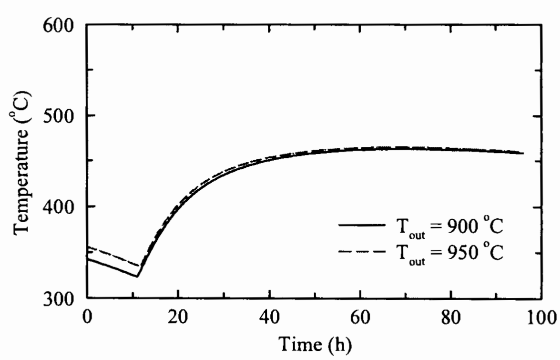

Figure 8. RELAP5-3D calculated maximum reactor vessel midwall temperatures during a depressurized conduction cooldown accident with the pebble-bed design.

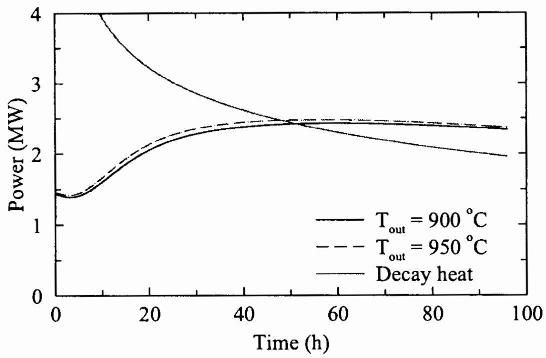

Figure 9. A comparison of RELAP5-3D calculated RCCS heat removal and core decay power during a depressurized conduction cooldown accident with the pebble-bed design.

Figure 10 shows axial temperature profiles at the inner surface of the reactor vessel at 0.0 and 83.3 h after the start of the accident. RELAP5-3D results are shown with symbols while the THERMIX results are shown with solid lines. The calculated temperature profiles were similar with both codes. At steady state, the temperature between the BAF and TAF was nearly independent of the elevation. The maximum temperature occurred below the outlet plenum, which was heated by the outlet flow but not cooled by flow through the riser and bypass flow paths as occurred at higher elevations. The maximum temperature during the conduction cooldown accident occurred near the elevation of the peak power location in the core, which was slightly above the core centerline fore this analysis. The trends were similar with both codes, but the maximum value obtained from RELAP5-3D was about 40 °C lower than the value from THERMIX.

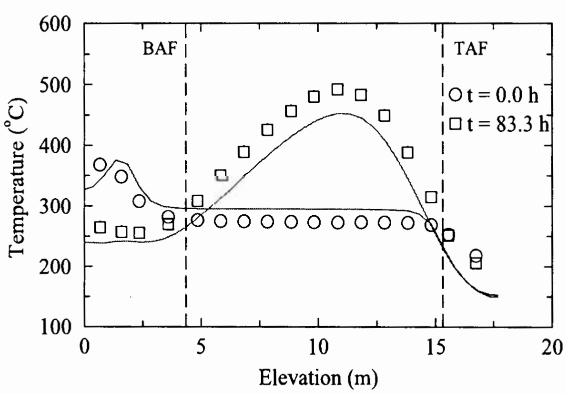

Figure 10. Reactor vessel axial temperature profiles for the pebble-bed design with an outlet fluid temperature of 950 °C.

The maximum temperatures calculated during the conduction cooldown event are given in Table 6. The 50 °C variation in initial outlet temperature had only a small effect on the maximum temperatures during the accident.

Table 6. Maximum temperatures during the depressurized conduction cooldown accident for the pebble-bed VHTR.

Parameter	T_{out} = 900 °C		T_{out} = 950 °C	
	R5-3D	THRMX	R5-3D	THRMX
Maximum vessel temperature (inner wall), °C	492	480	494	482
Maximum vessel temperature (midwall), °C	463	451	465	452
Maximum fuel temperature, °C	1486	1550	1493	1554

For the 950 °C outlet temperature case, the THERMIX calculation was allowed to run until the reactor pressure vessel temperature dropped below 371 °C, about 540 hours after shutdown (the control rods were assumed fully inserted to prevent re-criticality). The pressure vessel thus can be expected to remain at an elevated temperature for about 22 days assuming no active measures are taken to cool it. According to Table 1, SA-508 steel can withstand elevated temperatures for at least 40 days.

4.0 Conclusion

Current designs for the VHTR with a prismatic core do not allow the use of SA-508 steel for the reactor vessel. Best-estimate calculations of the maximum reactor vessel temperature during normal operation exceeded the value allowed for SA-508, but were at least 170 °C below the 590 °C limit for SA-336 when the design improvements described by Reza et al. were modeled. Note that the design of the important outlet plenum region has yet to be completed with the prismatic core.

For the pebble-bed design, the maximum vessel wall temperature during normal operation was 14 to 30 °C below the allowed value for unlimited operation of SA-508. Some design changes, such as the use of active vessel cooling, may be required to provide increased margin to the temperature limit during normal operation.

The maximum vessel temperatures increased significantly during a depressurized conduction cooldown accident. However, the temperatures remained at least 70 °C below the accident limits allowed by the ASME code for 1000 h of operation.

The PEBBED-THERMIX calculations of the pebble-bed VHTR were confirmed by RELAP5-3D calculations. During normal operation, the predicted axial profiles in reactor vessel temperature were similar with both codes and the predicted maximum values were within 2 °C. The trends of the calculated vessel temperatures were similar during the depressurized conduction cooldown accident. The maximum value predicted with RELAP5-3D was about 40 °C higher than that predicted by the THERMIX model using the same PEBBED heat source term. This agreement is considered reasonable based on the expected uncertainty in either calculation. The differences between the PEBBED-THERMIX and RELAP5-3D calculations were not large enough to affect conclusions concerning comparisons between calculated and allowed temperatures during normal operation and the depressurized conduction cooldown accident.

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